CP&L
A Progress Energy Company

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U. S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REVISED EXCESS FLOW CHECK VALVE TESTING FREQUENCY

Ladies and Gentlemen:

By letter dated January 17, 2001 (Serial: BSEP 00-0164), Carolina Power & Light (CP&L) Company submitted a license amendment application and Inservice Testing Program relief request for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, to revise the testing frequency for excess flow check valves (EFCVs). Additional information supporting the license amendment application and relief request was subsequently provided by letter dated March 23, 2001 (Serial: BSEP 01-0026). The purpose of this letter is to provide additional information, discussed with NRC representatives during a telephone call on August 16, 2001, clarifying CP&L's response to NRC Item 3 in the March 23, 2001, letter.

CP&L's response to NRC Item 3 in the March 23, 2001, letter confirmed that the operational impact of a postulated instrument line break has been assessed and that the operational impact will not be significant. This response also included analytical radiological results for orificed and non-orificed instrument line breaks to substantiate CP&L's assertion that the offsite radiological exposure from an unisolated instrument line break would be bounded by a main steam line break accident.

CP&L has compared exclusion area exposure data and mass release amounts for the main steam line break accident and the instrument line break and determined that the main steam line break accident bounds the instrument line break for Exclusion Area and Control Room exposure. The main steam line break accident scenario is considered the most limiting for Control Room exposure due to the high thyroid dose associated with the release. Data for the main steam line break accident was obtained from Updated Final Safety Analysis Report (UFSAR) Table 15.6.3-4 for offsite dose numbers and UFSAR Section 6.4.2.6 for Control Room dose. Mass release data for the main steam line break accident was obtained from UFSAR Table 15.6.3-2. The data for the exclusion area dose for the instrument line break was based on the analysis in the original FSAR, Section M14.3, Part 6. Data for mass released is obtained from this same analysis.

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A comparison of the exclusion area exposure for the main steam line break accident and the instrument line break is provided in the following table:

	TEDE	Thyroid
Main Steam Line Break	65.1 mR	793 mR
Instrument Line Break (Note 1)	0.1 mR	1.7 mR

Notes:

1. These results are applicable to a break of an instrument line equipped with a 0.25-inch flow-restricting orifice. These results assume operation of the Standby Gas Treatment system. Each BSEP instrument line connected to the reactor coolant pressure boundary is equipped with an excess flow check valve and a 0.25-inch flow-restricting orifice.

The Control Room dose is primarily a function of mass released. Mass released during a main steam line break accident would be 96,505 lbm over 10.5 seconds. Mass released from an instrument line break would be 30,000 lbm over 4 hours. The main steam line break releases to the Turbine Building without any filtration mechanism. The instrument line break releases to the Reactor Building and passes through the Standby Gas Treatment system, resulting in an elevated release via the plant's main stack. As a result of the lower mass released and the release flow path, it is logical to consider the Control Room exposure of the instrument line break to be less than that of the main steam line break accident. Instrument line breaks are not directly analyzed for impact to Control Room habitability (Standard Review Plan Section 15.6.2). The instrument line break was analyzed to meet the requirements of Safety Guide 11, "Instrument Lines Penetrating Primary Containment" (UFSAR 6.2.4.1.3). Doses to the public from both the main steam line break accident and the instrument line break are below 10 CFR Part 100 limits.

A copy of the UFSAR sections and the FSAR M14.3 section referenced above are enclosed.

Please refer any questions regarding this submittal to Mr. David C. DiCello, Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,

John S. Keenan

J. S. Keenan

WRM/wrm

Enclosure: Applicable UFSAR and FSAR Pages

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

M. G. Turbal

Notary (Seal)

My commission expires: May 18, 2003

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ENCLOSURE

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Contents

a) These lines, in general, are required to be operable during normal operation, shutdown, and accidents in order to monitor critical parameters. It is important to minimize the possibility of a failure of the isolation valve causing the instrumentation to be inoperable.

b) The instrument piping is constructed to a high level of quality and so the probability of failure is minimal, especially in light of its low pressure service.

c) The piping is leak tested during integrated containment leak rate tests except the local leak rate test is used for piping downstream of valves normally closed during the ILRT to verify its integrity and leak tightness.

d) There is negligible inventory loss as there is no direct leak from the Reactor Coolant Pressure Boundary. Isolation of the instrument line will not reduce the RPCB leakage.

e) Off-site dose calculations have been performed and the figures are significantly less than 10CFR100 limits.

Certain instrument lines are provided with two automatic isolation valves to meet the requirements of GDC-56. Certain instrument lines are considered an extension of primary containment up to the process instruments, since these instruments are required to be operable (i.e. not isolated) under normal, shutdown, and accident conditions to monitor critical parameters. Instrument valves that form a part of this boundary are considered manual containment isolation valves.

9) The TIP N_2 purge line is provided with a simple check valve outside containment. The off-site dose calculations for a break in this 0.375 inch line are significantly less than 10CFR100 limits. The combined effect of the small line and the check valve is adequate isolation for this line. Although this line is also provided with a normally open solenoid valve which will close on a high drywell pressure or low reactor water level, it is not considered a containment isolation valve because of the distance it is located from the containment.

10) The RHR head spray operating mode has been removed. The process line inside of the containment consists of a normally closed and electrically disabled motor operated valve, check valve, and blind flange, meeting the definition of a closed loop system inside of the containment.

6.2.4.1.3 Fluid instrumentation lines isolation criteria. The isolation provisions for fluid instrumentation lines that penetrate the primary containment were designed to meet the requirements of AEC Safety Guide 11. Certain instrument lines are considered an extension of primary containment up to the process instruments, since these instruments are required to be operable (i.e. not isolated) under normal, shutdown, and accident conditions to monitor critical parameters. Instrument valves that form a part of this boundary are considered manual containment isolation valves.

During either chlorine or radiation isolation, the mechanical equipment room was originally calculated to have an inleakage rate of 2640 cfm. 1760 cfm of this value resulted from a one square foot opening in an exterior wall which existed at the time of the original calculation. This hole has since been covered, such that the calculated value of 2640 cfm has a margin of 1760 cfm. Any air which enters the mechanical equipment room will be diluted by the air in the room, prior to being drawn into the ductwork. Therefore, this number is bounded by either the 3000 cfm for radiological calculations or the 2000 cfm (of non diluted outside air) for the chlorine tank car rupture calculation.

6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment. The following provisions were taken into consideration in the Control Room Area Ventilation System design to assure that there are no toxic or radioactive gases and other hazardous material that would transfer into the Control Room:

1. During radiation protection, smoke protection, and normal mode the Control Room envelope is maintained at static pressures slightly higher than atmospheric to prevent infiltration from the outside. During chlorine mode the Control Room is isolated but it is not pressurized.

2. Doors and other openings into the Control Room are conspicuously marked to assure that they will normally remain closed. This administrative control will assure that the Control Room normally remains closed. The doors are equipped with reclosers.

6.4.2.5 Shielding Design. The Control Room envelope is shielded against direct sources of radiation which are present during normal conditions and following a postulated accident.

There are no significant sources of direct or streaming radiation near the Control Room envelope during normal operating conditions.

The calculated whole body Control Room dose to operators during a postulated loss of coolant accident (LOCA) is 0.416 rem which is well within the current Nuclear Regulatory Commission (NRC) criteria of 5 rem given in Standard Review Plan 6.4. This dose is based on a Reactor Building concrete wall thickness of 2.0 ft and Control Building wall and roof concrete thickness of 2.0 ft each.

6.4.2.6 Doses Due to Main Steam Line Break Accident MSLB. The calculated radiological consequences of the BNP MSLB at a pre-accident iodine spike of 4.0 μCi/g I-131 equivalent is 16.2 rem, the thyroid dose (30 day exposure). Under the same initial conditions, in the chorine isolation mode the 30 day MSLB thyroid dose is 24.7 rem. Dose levels are below Nuclear Regulatory Commission (NRC) criteria of 30 rem thyroid given in Standard Review Plan 6.4.

Table 15.6.3-2

ASSUMPTIONS FOR MAIN STEAM LINE BREAK ACCIDENT (2558 MWt)

Power (MWt)		2558
Power Multiplier Factor		1.02
Initial Steam Dome Volume (ft^3)		4428
Iodine Concentration in Coolant (μCi/gm)		
Case 1:	I-131	0.073
	I-132	0.71
	I-133	0.50
	I-134	1.40
	I-135	0.73
	DEI-131	0.2
Case 2:	I-131	1.45
	I-132	14.1
	I-133	9.95
	I-134	27.9
	I-135	14.5
	DEI-131	4.0
MSIV Closure Time (sec)		10.5
Coolant Discharged from Break (lbm)		96,505
Fraction of Iodine in Released Coolant Assumed Airborne (%)		100
Noble Gas Release Rate prior to MSIV Closure (μCi/s at 30 min decay)		300,000
Holdup in Turbine Building		No
Release Height (m)		30
Meteorological Condition		Fumigation
K/Q at EA (sec/m^3)		8.4E-4
K/Q at LPZ (sec/m^3)		1.7E-4

Table 15.6.3-4

MAIN STEAM LINE BREAK ACCIDENT RADIOLOGICAL CONSEQUENCES
(2558 MWt)

LOCATION/TYPE	RATED POWER	UPRATE	LIMIT
Case 1: Iodine concentration in coolant = 0.2μCi/gm dose-equivalent I-131.			
Exclusion Area (2 hours):			
Whole Body Dose, rem	6.51E-2	6.51E-2	≤2.5
Thyroid Dose, rem	3.83	3.83	≤30
Low Population Zone (2 hours):			
Whole Body Dose, rem	1.35E-2	1.35E-2	≤2.5
Thyroid Dose, rem	0.793	0.793	≤30
Case 2:			
Iodine concentration in coolant = 4.0 μCi/gm dose-equivalent I-131.			
Exclusion Area (2 hours):			
Whole Body Dose, rem	1.25	1.25	≤25
Thyroid Dose, rem	76.6	76.6	≤300
Low Population Zone (2 hours):			
Whole Body Dose, rem	0.259	0.259	≤25
Thyroid Dose, rem	15.9	15.9	≤300

(6) Plant design with regard to a break in any of the instrument lines from the primary system that penetrate the containment are discussed in Subsections 5.2.3.5, H.5.21 and response to Comment M5.55(b).

Instrument lines connected to the reactor primary pressure boundary are typically 3/4-inch pipes which are provided with a 1/4-inch flow restricting orifice inside the primary containment and with an air-operated isolation valve outside the primary containment. This isolation valve is mounted as close as possible to the penetration, but in no case more than six inches from it.

A break inside the secondary containment should be readily detected by the loss of the given instrument channel and by secondary containment temperature and radiation monitoring instrumentation. A trip of the latter stops building ventilation and initiates the standby gas treatment system.

A break downstream of the isolation valve can be readily isolated. The concern here is over a break in the six inch (maximun) long line between the penetration and the isolation valve.

The potential effects of an instrument line break on the secondary containment and the resultant offsite doses from radioactivity released with the steam/water have been analyzed. The quantitative assessment of these effects using conservative assumptions show that they are less severe than those of the design basis accidents already analyzed for BSEP Units 1 and 2.

In the unlikely event of an instrument line break, the following effects could occur before corrective action can be completed:

a. Pressure and temperature transients in the secondary containment are expected to develop.

b. Humidity should increase.

c. Activity contained in the reactor coolant or main steam may be released through the break.

d. If the pressure transient is substantial, a portion of the activity could be released directly to the environs without the mitigating effect of the standby gas treatment system.

e. Radiation exposure may occur to the equipment in the secondary containment and, potentially, to people at the exclusion area boundary.

The analysis was done for two types of instrument lines - one containing saturated steam and the other saturated water. The saturated water case is the more severe case and is detailed below.

ASSUMPTIONS

1. The blowdown flowrates were computed according to Moody(*) for saturated water corresponding to the reactor pressure. The operating pressure in the reactor was assumed to be 1035 psig. The pressure remains constant for the first ten minutes and so does the blowdown flow rate. After ten minutes it is assumed that the operator initiates reactor shutdown, and reactor pressure decreases linearly to atmospheric in four hours. The flowrate and the enthalpy were computed corresponding to the reactor pressure and are shown in the following table.

Time (sec)	Mass Flowrate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	2.76	550.1
600	2.76	550.1
1200	2.69	544.1
1800	2.66	537.1
2600	2.59	528.3
4600	2.42	504.3
6600	2.25	477.9
8600	1.91	447.2
10600	1.50	409.8
15000	0.0	180.1

* Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture", Feb. 1965, Trans. of the ASME.

2. Each instrument line has a 1/4" restricting orifice.

3. The SGTS is started at one minute after the break occurs and has a constant flow rate of 3000 cfm; during the first minute, no credit is taken for heat removal through any ventilation system.

4. No credits have been taken for heat conduction to the internal structures or to the outside.

5. No frictional pressure drop is considered in the instrument line.

6. Initial conditions in the secondary containment are

Pressure, psia	=	14.7
Temperature, F	=	104
Humidity, %	=	90

7. Ambient pressure = 14.7 psia

The pressure and temperature transients were obtained from the computer code COMPRESS-QS, the results of which are shown below:

Peak Pressure, psig	none (negative pressure maintained)
Peak Temperature, F	127
Maximum Humidity, %	100

The results of the pressure transient analysis show that the structural integrity of the secondary containment will be maintained.

The resultant temperature of 127 F and relative humidity of 100 percent will not adversely affect the functional ability or performance of the SGTS. The SGTS moisture separator will remove any entrained moisture in the air entering the train. Following the moisture seperator, the electric heaters are designed to accept incoming 127 F air and to reduce the relative humidity from 100

percent to 70 percent. Experimental data does not indicate any significant adverse effect of this order of relative humidity on the iodine removal efficiency of the charcoal. The temperature of the stream entering the charcoal bed is well within the desorption and ignition temperature limit of the charcoal.

OFF-SITE DOSES

The off-site dose for the above transient was calculated using the following assumptions:

1. The redundant radiation monitors in the main ventilation system will provide a high radiation signal which results in closure of the main ventilation system and the actuation of the standby gas treatment system within one minute.

2. The release for the first one minute is through the roof vents, without any filtration; after that it is through the SGTS and up the plant stack (100 meters).

3. Atmospheric dilution for the first minute is based on a 30-meter release with fumigation conditions and no credit for building wake, in accordance with Regulatory Guide 1.5. Atmospheric dilution for the remaining period of the accident is that for elevated release (100 meters) in accordance with Regulatory Guide 1.3.

4. Reactor coolant activity corresponds to 500,000 Ci/sec of noble gases after 30-minutes decay.

5. Breathing rate is 3.47×10^{-4} m^3/sec.

6. SGTS filter efficiency is 90%.

7. All isotopes of iodine are considered.

The resulting dose is 1.7 mrem to the thyroid. The whole body dose is much less than 0.1 mrem.